September 22, 2003

MEMORANDUM TO: Stuart Richards, Chief
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SUBJECT: RESULTS OF THE CRYSTAL RIVER SDP PHASE 2 NOTEBOOK BENCHMARKING VISIT

During February 2003, NRC staff and contractors visited the Crystal River Nuclear Plant Unit 3 (CR-3) in Crystal River, Florida to compare the CR-3 Significance Determination Process (SDP) Phase 2 notebook and licensee's risk model results to ensure that the SDP notebook was generally conservative. The current plant probabilistic risk assessment's (PRA's) internal event core damage frequency was 6.39E-6/reactor-year excluding internal flood events. The CR-3 PRA did not include an integrated PRA model with external initiating events. Therefore sensitivity studies were not performed to determine any impact of external initiators on SDP color determinations. In addition, the results from analyses using the NRC's draft Revision 3i Standard Plant Analysis Risk (SPAR) model for CR-3 were also compared with the licensee's risk model. The results of the SPAR model benchmarking effort will be documented in the next revision of the SPAR (revision 3) model documentation.

In the review of the CR-3 SDP notebook for the benchmark efforts, the team determined that some changes to the SDP notebook were needed to reflect how CR-3 is currently designed and operated. Fifty two hypothetical inspection findings were processed through the SDP notebook and compared with the licensee's related importance measures. Using the Revision 0 SDP notebook, the team determined that 16.5 percent of the cases were less conservative, 34.5 percent of the cases were more conservative, and 49 percent of the cases were consistent with the licensee's results. Of the conservative cases, 8 cases were two or more colors greater than the results obtained using the licensee's model. Consequently, 32 changes were made to the SDP notebook.

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Using the Revision 1SDP notebook, the team determined that 2 percent of the cases were less conservative, 27 percent of the cases were more conservative, and 71 percent of the cases were consistent with the licensee's results. Of the conservative cases, all but 2 cases were one order of magnitude greater than the results obtained with the licensee's model and as such were generally consistent with the expectation that the notebooks should be slightly conservative when compared with the licensee's model.

The licensee's PRA staff had substantial knowledge of both the CR-3 PRA model and conduct of plant operations. The licensee's comments greatly improved the quality and content of the SDP notebook.

Attachment A describes the process and specific results of the comparison of the CR-3 SDP Phase 2 Notebook and the licensee's PRA.

Attachment: As stated

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SUMMARY REPORT ON BENCHMARKING TRIP

TO THE CRYSTAL RIVER NUCLEAR PLANT UNIT 3

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ATTACHMENT A

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1. INTRODUCTION

A benchmarking of the Risk-Informed Inspection Notebook for the Crystal River Nuclear Plant (CRNP) was conducted during a plant site visit on February 11-13, 2003. NRC staff (R. Bernhard, M. Franovich, and W. Rogers) and BNL staff (P. Samanta) participated in this Benchmarking exercise.

In preparation for the meeting, BNL staff reviewed the SDP notebook for the Crystal River Nuclear Plant and evaluated a set of hypothetical inspection findings using the Rev. 0 SDP worksheets. In addition, NRC staff provided the licensee with a copy of the meeting protocol.

The major milestones achieved during this meeting were as follows:

1. Recent modifications made to the Crystal River PRA were discussed for consideration in the Rev. 1 model to be prepared following benchmarking.

2. Importance measures, including the Risk Achievement Worths (RAWs) for the basic events in the internal event model for average maintenance, were obtained from the licensee.

3. Benchmarking was conducted using the Rev. 0 SDP model and the revised SDP model considering the licensee's input and other modifications that were judged necessary based on comparison of the SDP model and the licensee's detailed model.

4. For cases where the color evaluated by the SDP notebook differed from that determined based on the RAW values generated by the updated licensee's PRA, results of the licensee's model including the detailed minimal cutsets were requested from the licensee. The cutsets were reviewed to understand the reasons for the differences. Applicable changes were defined for the SDP model.

2. SUMMARY RESULTS FROM BENCHMARKING

Summary of Benchmarking Results

Benchmarking of the SDP Notebook for the Crystal River Nuclear Plant showed consistency of the significance of the inspection findings obtained using the notebook with that to be obtained using the plant PRA. As expected, in some cases, the results of the notebook are conservative. Two cases of a conservative result by two orders of magnitude (i.e., the significance obtained using the notebook is two colors higher than that to be obtained using the plant PRA) and one case where the notebook provides an underestimation were noted. A summary of the results of the risk characterization of hypothetical inspection findings is as follows (51 of 52 cases were comparable to the licensee's model):

2% (1 of 51 cases)	underestimation of risk significance
4% (2 of 51 cases)	overestimation of risk significance by two orders of magnitude
23% (12 of 51 cases)	overestimation by one order of magnitude
71% (36 of 51 cases)	consistent risk significance.

Detailed results of benchmarking are summarized in Table 1. Table 1 consists of eight columns. The first and second columns identify the components or the case runs. The assigned colors from the SDP Rev. 0 worksheets without incorporating any modification from the benchmarking exercise are shown in the third column. The fourth column gives the basic event name in the plant PRA used to obtain the risk achievement worth (RAW) for the component out of service or the failed operator action. The fifth and sixth columns respectively show the licensee's internal RAW value and the color to be defined based on the RAW values from the latest PRA model. The seventh column presents the colors for the inspection findings based on Rev. 1 version of the notebook. The Rev. 1 version of the notebook is prepared considering the revisions to the Rev. 0 version of the SDP notebook judged applicable during benchmarking. The last column provides comments identifying the difference in results between the SDP Rev. 1 notebook and the plant PRA, and the applicable rules in obtaining the color of the inspection finding using the SDP notebook.

Table 2 presents a summary of the comparisons between the results obtained using the Crystal River Nuclear Plant SDP Notebook and the plant PRA. It also shows a comparison of the results using Rev. 0 and Rev. 1 versions of the notebook. The results show that overestimations by the notebook were reduced and the matches were increased through revisions to the notebook implemented as a result of benchmarking. The matches increased from 49% to 71%. Non-conservative results decreased from 16.5% to 2% and conservative results by two or more orders of magnitude decreased from 16.5% to 4%.

Discussion of Non-conservative results by the notebook

One case of underestimation or non-conservative result related to an operator action was noted during the benchmarking. The reason for the underestimation is summarized as follows:

- An underestimation by one color was obtained for the operator failure to refill the BWST. The licensee obtains a risk significance of white where the notebook assesses a significance of Green. Differences in Steam Generator Tube Rupture (SGTR) frequency is attributed to this non-conservatism. In the notebook, SGTR is assigned to Row III, representing a frequency of 1E-3. The plant PRA had a frequency of 3E-3 for each Steam Generator representing a frequency of approximately 6E-3.

<u>Discussion of conservative results by the notebook</u>

Fifteen cases of overestimation (two cases by two colors and thirteen cases by one color) were noted during benchmarking. Overestimation by two colors was analyzed and is discussed below. Some observations are made regarding the overestimations by one color.

1. The risk significance of Makeup pump 1B is overestimated by two colors by the notebook compared to the plant PRA. The plant PRA estimates a Green significance whereas the notebook estimates a Yellow significance. The difference is attributed to the differences in the pump failure probability in the PRA vs the credit given in the notebook, and the differences in the SLOCA and MLOCA initiator frequencies between the plant PRA and the notebook.

2. The risk significance of DC Panel DPDP-5A is overestimated by two colors compared to the plant PRA. DPDP-5A supports the makeup pump A which contributes to the high significance for the panel in the notebook evaluation. As noted above, the risk significance of the makeup pumps are estimated to be low in the plant PRA.

In general, the reason for the conservative results by the notebook compared to the plant PRA can be attributed to the differences in initiating event frequencies between the plant PRA and the notebook, and differences in component unavailability, particularly for the makeup pumps, as discussed above. The initiator frequencies for LLOCA, MLOCA, SLOCA, and ATWS are lower in the PRA compared to the values used in the notebook. LLOCA, MLOCA, and SLOCA initiating frequencies in the plant PRA are respectively 5.0E-6, 4.0E-5, and 5.0E-4. The corresponding frequencies assumed in the notebook are 1E-5, 1E-4, and 1E-3. The ATWS frequency in the PRA is less than 1E-7, and it is assumed to be 1E-6 in the notebook.

<u>Changes incorporated following benchmarking resulting in updating of benchmarking results</u>

Following benchmarking, some changes were decided based on further analysis of the information gathered during benchmarking. The important changes made to the notebook can be summarized as follows:

1. Loss of Startup/backup transformer (LEST) initiating event was modeled. Worksheets and event trees are included.

2. SLOCA and SORV worksheets and event trees are modified to include the need for operator action to either raise the SG level or open the PORV.

3. Auxiliary Feedwater pump FWP-7 and MDEFW pump credits were adjusted to be consistent with the plant HEPs.

4. Stuck-open safety relief valve (SOSRV) credit was restored to “1 train” from a credit of 1.

5. LEAC worksheet was deleted.

These changes resulted in better correspondence with the plant PRA. In addition, three additional cases were evaluated (EDG B, operator failure to crosstie 4 kV ES bus, and operator failure to raise SG level in SLOCA).

Table 1: Summary of Benchmarking Results for Crystal River Nuclear Plant, Unit 3

Internal Events CDF is 6.39E-06/reactor-year, excluding internal flooding
RAW Thresholds are W = 1.16, Y = 2.57, and R = 16.65
Truncation Level at 1E-10

No.	***Component Out-of-Service or Failed Operator Action***	***SDP Worksheet Results (Before)***	***Basic Event Name***	***Internal RAW***	***Site Color***	***SDP Worksheet Results (After)***	***Comments***
1.	Core flood tank	Y	Truncated	~1.0	G	G	
2.	EDG-3A FTS	W	ADGES3AA	1.53	W	W	
3.	EDG-3B FTS	W	ADGES3BA	2.9	Y	Y	
4.	Battery DPBA-1A	R	DMMBT1AF	1.35	W	W	Battery charger can carry the SI loads, LOOP and LEST worksheets are evaluated.
5.	DC bus DPDP-5B	R	DMMDP5BF	12.83	Y	R	Conservative by one order.
6.	DC bus DPDP-5A	R	DMMDP5AF	2.20	W	R	Conservative by two orders of magnitude.
7.	Loss of non 1E ‘C’ bus	Not modeled	IE_T14	29.43	R	R	
8.	4 kV vital bus 3A	R	AB24KEAF	14.58	Y	R	Conservative by one order of magnitude.
9.	EFP-1 motor-driven pump FTR	G	Truncated	~1.0	G	W	Conservative by one order of magnitude.
10.	EFP-2 turbine-driven pump FTR	G	QMMEFP2F	1.44	W	W	
11.	EFP-3 Diesel-driven pump FTR	G	QMMEFP3F	2.05	W	Y	Conservative by one order of magnitude.

No.	Component Out-of-Service or Failed Operator Action	SDP Worksheet Results (Before)	Basic Event Name	Internal RAW	Site Color	SDP Worksheet Results (After)	Comments
12.	FWP-7 (aux feed) FTR (mechanical failure)	G	QMMFWP7F	1.22	W	W	
13.	FWP-7 misaligned	G	QMMEFP7X	1.22	W	W	
14.	MTDG-1 FTR (support DG for FWP-7)	G	ADGMTDGF	1.09	G	G	LOOP and LEST worksheets were solved since FWP-7 is normally powered from non-vital power.
15.	1NSCCC pump FTR (SWP 1A)	G	SPMSWPAA F	5.75	Y	Y	
16.	DHCCC Train B faults	Y	SMMDHCCB	10.01	Y	Y	
17.	DHSW Train (RWP -3B)	Y	SMMRW3BF	9.99	Y	Y	
18.	1 DHSW pump (RWP-3B)	Y	SMMRW3BF	9.99	Y	Y	
19.	PCS/MFW initiator	G	IE_T2	1.04	G	G	
20.	1 SSCCC pump FTR	G	Truncated	~1.0	G	G	
21.	(HPI) MUP-1B pump B FTR	Y	HPM001BF	1.13	G	Y	Conservative by two orders of magnitude. (1)
22.	(HPI) MUP-1C FTR	Y	HPM001CA	1.73	W	Y	Conservative by one order. (1)
23.	Air Compressor FTS	G	Truncated	~1.0 (2)	G	G	
24.	PORV RCV-10 FTO (main valve, not pilot)	W	RRVRC10N	4.37	Y	Y	
25.	PORV FTC (SORV)	Y	RRVRC10C	1.21	W	Y	Conservative by one order.
26.	PORV Block Valve RCV-11 FTC	W	RMVRC11C	1.58	W	W	
27.	1 RHR/LPI Pump FTR	Y	LMMDHPAF	9.93	Y	Y	

No.	Component Out-of-Service or Failed Operator Action	SDP Worksheet Results (Before)	Basic Event Name	Internal RAW	Site Color	SDP Worksheet Results (After)	Comments
28.	1 RHR HX plugged	Y	LPM001AM	6.68	Y	Y	
29.	Failure of Cooling to AHHE-30A to DCP-1A	Y	SMM3238X	3.20	Y	Y	
30.	DCP-1A &1B CCF	R	SMMDCCCF	146.02	R	R	
31.	Reactor Building Sump valve FTO DHV-42	Y	LMMDV42F	9.90	Y	Y	
32.	ECCS piggyback valve (DHV-11)	Y	LMMDV11F	9.84	Y	Y	
33.	DHV-3 FTO (RCS hotleg dropline valve)	R	LMMDHRSF	1.09	G	W	Conservative by one order.
34.	BWST level transmitters CCF (BWST fails)	R	LTKBWSTJ	79.25	R	R	
35.	One MSIV FTC (MSV-411)	Y	PAVM411C	1.03	G	Y	Licensee's RAW is not comparable. Licensee does not model pressurized thermal shock concerns, which are included in the SDP model.
36.	OTSG ADV FTO (MSV-25)	G	not found	~1.0	G	W	Conservative by one order.
37.	AMSAC	W	Truncated	~1.0	G	G	
38.	1 primary SRV FTO (RCV-8)	W	RRVRCV8N	1.17	W	W	
39.	1 primary SRV FTC	R	RMMRCVSC	7.70	Y	Y	
	Operator Action						
40.	Restore or use main feedwater	G	PHURMFWR	1.0	G	G	
41.	Refill BWST	Y	WHUBWSTY	1.28	W	G	Non-conservative by one order.
No.	**Component Out-of-Service or Failed Operator Action**	**SDP Worksheet Results (Before)**	**Basic Event Name**	**Internal RAW**	**Site Color**	**SDP Worksheet Results (After)**	**Comments**

42.	Start FWP-7	G	QHUFWP7Y	3.39	Y	Y [4)]	To properly compare, the X4KV crosstie is also failed based on examination of the cutsets. This accounts for the operator action dependencies
43.	Feed & bleed	W	RHUPORVY[(3)]	1.07	G	G	To properly compare to the basic evnt RAW, SGTR and SLOCA affected sequences are solved.
44.	Close PORV block valve	W	not found	1.58	W	W	Use RAW for Block Valve FTC.
45.	Initiate high pressure recirculation	R	HHUHPRCY	155.51	R	R	
46.	Initiate low pressure recirculation	Y	LHULPRCY	1.76	W	Y	Conservative by one order.
47.	Emergency borate during ATWS	Y	HHUMANUZ	1.0	G	W	Conservative by one order.
48.	Realignment of DHCCC cooling for HPI pump A on loss of NSCCC	G	SHUMADCY	1.03	G	G	
49.	Depressurizes the RCS & initiates decay heat removal (DHR) mode during SGTR	R	RHUCOOLY	71.86	R	R	
50.	Trip RCPs on loss of cooling	Y	RHURCPTY	1.03	G	W	Conservative by one order.
51.	Crosstie 4 kV ES Bus	NA	AHU4KVXY	1.39	W	Y	Conservative by one order.
52.	Operator fails to raise SG level in a SLOCA	NA	QHUEFW9Y	13.18	Y	Y	

Notes:

1. RAW values for MUP-1B and 1C, including internal flooding, were respectively 5.56 and 2.62.

2. In the licensee's PRA, the basic event, MPSI018H (IA-18-PS fails high) has a RAW of 7.37. This is an error in the licensee's model. PS-18 only controls IAP-4 diesel air compressor.

3. This basic event corresponds to operator opening PORV for pressure relief, as modeled in the SLOCA and SGTR worksheets. The licensee does not model feed and bleed as an operator action for transients. The function is considered automatic. For comparison purposes, the SDP evaluation is conducted using only the SLOCA and SGTR worksheets resulting in a Green finding.

4. SDP evaluation is conducted by failing the FWP-7 and failing to crosstie 4 kV ES Bus In the LEST worksheet. In the licensee's PRA, these two actions are considered dependent and following failure of FWP-7 the HEP for failure to crosstie 4 kV Bus is assumed to be 1.0.

Table 2: Comparative Summary of the Benchmarking Results

Cases Compared		SDP Notebook Before (draft Rev. 1)		SDP Notebook After (Rev. 1)	
		Number of Cases (52)	Percentage	Number of Cases (52)	Percentage
SDP: Less Conservative		8[1]	16.5	1	2
SDP: More Conservative	one order	9	17	12	23
	two orders	7[2]	16.5	2	4
SDP: Matched		25	49	36	71
Not modeled or comparable RAW not available		3		1	

Notes:

1. Two cases were non-conservative by two orders of magnitude.
2. One case was conservative by three orders of magnitude.

3. PROPOSED MODIFICATIONS TO REV. 0 SDP NOTEBOOK

A set of modifications are proposed for the Rev. 0 SDP notebook as a result of the site visit. These proposed modifications are driven by the licensee's revisions to the plant's PRA, better understanding of the current plant design features, revised Human Error Probabilities (HEPs), modified initiator frequencies, and the results of benchmarking.

3.1 Specific Changes to the Rev. 0 SDP Notebook for the Crystal River Nuclear Plant

The following changes were made based on the licensee inputs and evaluations conducted during and following benchmarking:

<u>Summary of changes following benchmarking to prepare the Rev. 1 version of the notebook</u>

1. Changes to Table 1

1.1 Loss of Startup/Backup ES Transformer event was added to Row I.

1.2 Loss of Non-safety DC Bus C event was added to Row III.

1.3 LOOP with one EDG available (LEAC) was deleted.

2. Changes to Table 2

2.1 Support system for diesel-driven EFW pump was modified to include local DC and self-contained auto-start. Support system for MDEFWP was changed to safety AC (from non-safety AC). Dependency on EFIC was added for the diesel-driven and the turbine-driven EFW pumps. A footnote is added that common cause failure of EFIC will cause failure EFP-1, EFP-2, and EFP-3.

2.2 Support system for the AFW pump (FWP-7) was revised to include non-safety AC, AOV flow control valves with own IA compressor FWP-9, condensate storage tank CDT-1 supply, and non-safety dedicated diesel (MTDG-1) backup. A footnote is added that non-safety DC Bus C is needed for starting MTDG-1.

2.3 DHCCC was noted as the support system for LPI/DHR. Previously, DHCCC was noted as the support system only for the heat exhangers.

2.4 It is noted that NSCCC is backup for Makeup pump C.

2.5 AC dependency for EDG was removed. Separate rows for EDG A and B are defined to address the plant alignment (to EDG B) for loss of startup/backup transformer. It is footnoted that the diesels are air cooled and a third, non-safety diesel is planned to be installed.

2.6 ESAS is included as a support system for NSCCC.

2.7 Dependency of IA compressors on non-safety AC and DC was noted.

2.8 It is clarified that the dependency for the steam-line ADV valves is primarily on bottled air backed up by IA.

2.9 Circ water system was noted as the support system for SSCCC.

2.10 It is noted that PORV pilot valve is supported by non-vital DC and the main valve is supported by DC Bus A.

2.11 BWST refill was added in Table 2 as a separate row.

2.12 A footnote is added that the failure of automatic turbine trip should be referred to the senior reactor analyst or risk analyst because of its high RAW value.

2.13 Initiating event scenario column was revised based on the changes made as part of the benchmarking lessons.

3. Changes to worksheets and event trees

3.1 Operator action credit for aligning the feedwater trains, defined as the PCS function, was changed to 1 based on the HEP, in applicable worksheets.

3.2 Operator action credit of 1 was assigned for aligning FWP-7 pump. Similarly, an operator action credit of 1 is assigned for MDEFWP. This implied a combined credit of 2 for aligning FWP-7 and MDEFWP. No credit was given if only alignment of MDEFWP was involved since the associated HEP is 0.5.

3.3 Mitigation capability for High Pressure Injection was defined as 1/2 Makeup/HPI trains or operator aligns the third pump with a credit of 1 multi-train system.

3.4 In the TPCS worksheets and other transients with loss of the power conversion system, emergency feedwater function was split into automatic and manual actions. Failure of the automatic action increases the possibility of a stuck-open relief valve. Stuck-open safety relief valve (SOSRV) is explicitly modeled in defining the accident sequences. Both steam and liquid relief are possible. TPCS worksheet and event tree were modified addressing these changes.

3.5 In the SLOCA worksheet, operator action to maintain or raise SG level was modeled. Worksheet and event tree were modified to represent that the failure of this action coupled with the failure to open PORV for pressure relief results in core damage.

3.6 In the MLOCA worksheet, the operator action credit for HPR was changed to 3 (from 2).

3.7 In the LLOCA worksheet, the mitigation capability for Core Flood was changed to 1/2 core flood tanks (1 multi-train system).

3.8 The LOOP worksheet and event tree were revised to address the following changes. Emergency feedwater function was divided into automatic and manual action for situations where one of the EDGs operate. For station blackout scenarios, it is divided into emergency feedwater-diesel (EFWD) and emergency

feedwater-turbine (EFWT). For continued operation of the turbine-driven pump, recovery of offsite power in 4 hours is needed because of battery depletion. Seal LOCA resulting from failure of the operator to close the bleed-off line was also modeled. No operator action credit for recovering offsite power within 1 hour was given. Feed and bleed was modeled to be accomplished using 1/2 SRVs since the PORV is assumed lost due to loss of the PORV pilot valve as a result of LOOP.

3.9 Steam Generator Tube Rupture (SGTR) worksheet and event tree were modified using the revised modeling in the plant PRA. The revised tree is consistent with the modeling approach used for the SDP notebooks. Operator action credits were also modified using the revised plant-specific HEPs.

3.10 ATWS worksheet and event tree were modified to remove credit for PCS and to include use of AMSAC for turbine trip.

3.11 For modeling loss of Nuclear Service Closed Cycle Cooling System (NSCCC), mitigation credit for secondary heat removal was limited to operator action equal to 3 because loss of EFIC can result from the loss of HVAC resulting from the failure to start the dedicated chillers. Loss of EFIC will also result in loss of the feedwater pumps, and accordingly feedwater was not credited. Diesel-driven, turbine-driven, and the AFW pump (FWP-7) were credited, but limited by the operator action discussed above. For high pressure injection function, operator needs to start pump C or restore cooling to pump A. A combined credit of operator action=3 was assigned.

3.12 LEAC worksheet and event tree were deleted.

3.13 MSLB worksheet and event tree were revised considering the standard approach used in SDP notebooks. Concerns for pressurized thermal shocks are modeled. This modeling differs from the modeling in the plant PRA.

3.14 Loss of Non-Safety DC Bus C was modeled. Worksheet and event tree were added. The loss of non-safety DC Bus C results in a plant trip without a direct turbine trip, subsequent overcooling and ES actuation, but unavailability of main feedwater and RCS PORV. This event also restricts the ability to recover MFW and start auxiliary feedwater pump FWP-7.

3.15 LOIA worksheet and event tree were modified similar to the modifications made for transients without power conversion system (TPCS).

3.16 Worksheet and event tree were added for the special initiator “Loss of Startup/Backup ES Transformer (LEST)”. At the Crystal River plant, loss of startup transformer (SUT) will result in loss of the RCPs and MFW. Since the backup ES transformer (BEST) is arranged in parallel with the SUT, it is assumed that BEST will also fail. This situation creates a demand for EDG B. LEST frequency for the plant was 1.18E-1/year.

3.2 Generic Change in 0609 for Inspectors

None identified.

3.3 Generic Change to the SDP Notebook

In the Crystal River notebook, an operator action was considered for maintaining the SG level or opening a PORV in a SLOCA. Failure of this action is assumed to lead to a core damage. Applicability of such action for other B&W plants may be considered.

4. DISCUSSION ON EXTERNAL EVENTS

Integrated external event PRA model was not available for the Crystal River plant. No evaluation was conducted for the external event risk during the benchmarking exercise.

5. LIST OF PARTICIPANTS

Rudy Bernhard	USNRC - Region II
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